NON-PUBLIC?: N
ACCESSION #: 9206230187
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Salem Generating Station - Unit 2 PAGE: 1 OF 06

DOCKET NUMBER: 05000311

TITLE: Reactor Trip On No. 23 S/G Low-Low Level
EVENT DATE: 05/14/92 LER #: 92-009-00 REPORT DATE: 06/12/92

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 015

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: M. J. Pollack - LER Coordinator TELEPHONE: (609) 339-2022

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On 5/14/92, at 1801 hours, during a plant startup, a reactor trip on No. 23 Steam Generator (S/G) Low-Low Level occurred from 15% reactor power. Prior to the trip, the No. 23 S/G had experienced level control problems.

Two control modules required replacement. To support replacement of the modules, the No. 23 S/G main feedwater control valve (23BF19) was closed leaving mass inventory control via the No. 23 S/G feedwater control valve bypass valve (23BF40). However, contrary to this understanding, the 23BF19 valve remains open approximately 0.25" when at 20% reactor power; therefore, when the module replacement work was initiated and the 23BF19 valve demand was reduced to 0%, in preparation of 23BF19 valve control return, it fully closed. This eventually lead to the reduction of S/G mass inventory to below the reactor trip setpoint (16%). The root cause of this event is "Management/Quality Assurance Deficiency". Operations personnel understanding of the BF40 valve (4" Globe Valves) capability for maintaining S/G mass inventory (independent of the BF19 valves) was

not correct. Operations procedure, "Hot Standby to Minimum Load" has been revised. The circumstances surrounding this event will be reviewed with applicable Operations Department personnel and will be reviewed by the Nuclear Training Center. PSE&G will review its past response to INPO Significant Operating Experience Report No. 84-4. The BF19 and BF40 valves position indication will be reviewed by engineering.

END OF ABSTRACT

PLANT AND SYSTEM IDENTIFICATION:

Westinghouse - Pressurized Water Reactor

Energy Industry Identification System (EIIS) codes are identified in the text as {xx}

IDENTIFICATION OF OCCURRENCE:

Reactor Trip on No. 23 Steam Generator Low-Low Level

Event Date: 5/14/92

Report Date: 6/12/92

This report was initiated by Incident Report No. 92-319.

CONDITIONS PRIOR TO OCCURRENCE:

Mode 1 Reactor Power 20% - Unit Load 0 MWe

DESCRIPTION OF OCCURRENCE:

On May 14, 1992, at 1801 hours, a reactor trip on No. 23 Steam Generator (S/G) Low-Low Level occurred from 15% reactor power, during a plant startup. The plant was being returned to service after being taken off line (at 0010 hours that day) due to high S/G cation conductivity.

Prior to the trip, the No. 23 S/G was experiencing level control problems. Twice that day, No. 23 S/G level reached a low level of approximately 19%. The reactor trip low-low level setpoint is 16%. Subsequently, Instrumentation & Controls (I&C) technicians identified that the level error controller and feedwater flow control modules were showing sensitivity to AC noise.

To support replacement of the modules, the No. 23 S/G main feedwater control valve (23BF19) was to be fully closed, feedwater pump differential pressure increased and the No. 23 S/G feedwater control valve bypass valve (23BF40) throttled via operator control.

The decision to make repairs at 20% reactor power was made: 1) to avoid additional level perturbations (with a substantial reactor trip risk) if reactor power were to be reduced and 2) based on operations personnel belief that the 23BF40 valve (a 4" Masoneillan globe valve) would be able to maintain S/G level at 20% power. I&C personnel had informed Operations that the 23BF19 control would be unavailable during module replacement.

The sequence of events leading to the reactor trip included:

1. The 23BF19 and 23BF40 valves were in automatic control with

DESCRIPTION OF OCCURRENCE: (cont'd)

S/G level at 33%.

2. The 23BF19 controller was taken to manual and 23 S/G level raised to 44%.

3. Steam Generator Feedwater Pump (SGFP) differential pressure was increased slightly and 23BF19 demand was reduced to 31%. The Nuclear Control Operator (NCO), stationed at the 23BF19 and 23BF40 valves, informed the Control Room that the BF19 had closed.

The 23BF40 valve had moved from its full open position and appeared to control feed flow.

4. The 23BF40 valve control was placed in manual. At this point, the 23BF40 valve closed an additional 0.5" due to valve overstroke.

5. The I&C module changeout began.

6. Operations personnel initiated preparations for BF19/40 level control return by reducing 23BF19 demand to zero. At this time, the 23BF19 valve was observed to move in the

closed direction by the NCO.

7. A 23 S/G level swell phenomenon (a momentary increase in level) was experienced, level went from 44% to 48%. Operations personnel slightly closed the 23BF40 valve in response to the rise.

Following the swell, indicated level reduced to approximately 30%.

8. Reactor power was reduced to 15% causing T sub avg to decrease resulting in additional level shrink phenomenon (a momentary decrease in level); narrow range level indication showed 22%.

9. I&C informed Operations that 23BF19 control was back in service; the Operator opened the 23BF19 valve resulting in additional level shrink.

10. Per design, the reactor tripped when No. 23 S/G level reached 16%. Indicated level had decreased to 3%.

On May 14, 1992, at 1841 hours, the Nuclear Regulatory Commission was notified of the reactor trip on No. 23 S/G low-low level in accordance with Code of Federal Regulations 10CFR 50.72 (b)(2)(ii).

Following the reactor trip, a reactor cooldown occurred. In

DESCRIPTION OF OCCURRENCE: (cont'd)

accordance with Emergency Operating Procedure EOP-TRIP-2, a Main Steamline Isolation (an ESF) was initiated stopping the cooldown. The plant was stabilized in Mode 3 (Hot Standby).

APPARENT CAUSE OF OCCURRENCE:

The root cause of this event is "Management/Quality Assurance Deficiency" per NUREG 1022, "Licensee Event Report System" guidance.

Operations personnel understanding of the BF40 valve capability for controlling level (independent of the BF19 valves) was not correct. This misunderstanding had occurred due to: 1) incorrect assessment of 23BF40 capability test results conducted earlier that day and 2) information provided in the S/G water level control training system description which is used by operators as a guidance document during training. The training system description indicated that in the full open position, the BF40 valve will pass up to 25% of rated feedwater flow (however no qualifying information is provided).

The reduction in No. 23 S/G mass inventory, per Engineering analysis, was the result of the 23BF19 valve fully closing when the 23BF19 valve demand was reduced to 0% in preparation of 23BF19 valve control return. The 23BF19 valve initially caused the observed swell phenomenon followed by the reduction of S/G mass inventory. Engineering analysis shows that this event by itself could reduce level to below the trip setpoint. Simulator testing has shown that the BF40 valves will not accommodate adequate flow to the S/Gs at 20% reactor power even with a Steam Generator Feedwater Pump differential pressure of 118 psid. During event investigation, it was later found that the PSE&G Configuration Baseline Document (CBD) states that the valves are only capable of maintaining S/G mass inventory to between 7% and 8% reactor power.

Prior to Operations authorizing the module replacement, testing was conducted to determine if the 23BF19 valve was fully closed at 20% power and feedwater flow maintained. During the test, the 23BF19 valve was determined to be closed when it had stopped moving and the 23BF40 valve started moving (closed). Post event investigation determined that the 23BF19 is open approximately 0.25" at 20% power.

The 23BF40 observed movement is due to the correction of initial valve overstroke.

At the start of the sequence of events, it was again assessed that the 23BF19 valve had closed fully based on it stopping and the 23BF40 valve began to close.

Other factors which contributed to the cause of this event include:

1. Valve position indication is not provided in the control room and local indication is provided by a demarked metal plate with a pointer (approximately 0.5 inch from the

APPARENT CAUSE OF OCCURRENCE: (cont'd)

plate).

2. Recovery from the swell phenomenon included operator

actions which caused level shrink phenomenon. These actions included slightly closing the 23BF40 valve and reduction of reactor power. These actions did not significantly contribute to the reduction in S/G level as per Engineering analysis.

3. Just prior to the reactor trip, the 23BF19 valve was opened after I&C informed Operations that valve control was returned. The addition of cold water caused additional level shrink phenomena.

ANALYSIS OF OCCURRENCE:

The Low-Low S/G Level reactor trip prevents operation with S/G water level below the minimum required for adequate heat removal. The trip occurs on two out of three low-low level signals in any S/G. The setpoint ensures adequate S/G inventory, at the time of a reactor trip, to allow for possible Auxiliary Feedwater Pump starting delays; thus preventing S/G dryout and Reactor Coolant System {AB} thermal and hydraulic transients associated with a loss of the heat sink.

The BF19 valves regulate main feedwater flow to the S/Gs during normal power operation. During startup and low power operation the BF40 valves, in conjunction with the BF19 valves, regulate main feedwater flow.

The Reactor Protection System (RPS) {JC} functioned as designed and the heat sink was maintained during this event. Since the RPS is designed for the thermal and hydraulic effects of four-hundred (400) full power reactor trips, this low power reactor trip resulted in a thermal transient well within the design limits of the system. This event therefore involved no undue risk to the health or safety of the public; however due to the RPS system actuation and the initiation of Main Steamline Isolation, the circumstances surrounding these events are reportable in accordance with Code of Federal Regulations 10CFR 50.73 (a)(2)(iv).

The "shrink" and "swell" phenomena, discussed in the Apparent Cause of Occurrence section, occur following level transients. The initial S/G level response is opposite that of where S/G level stabilizes. Examples of "shrink" during this event include reduction of reactor power (causing reduction of T sub avg) and the addition of "cold" feedwater (when the 23BF19 was opened).

Investigation of the reactor trip identified a failure of the 21TB10

Turbine Bypass System valve. It failed open, when its feedback positioner arm broke. Analysis of the sequence of events leading to

ANALYSIS OF OCCURRENCE: (cont'd)

the reactor trip show that had it failed during the event, it would not have significantly contributed to the reduction in No. 23 S/G level based on valve design basis information.

The reduction in T sub avg, requiring main steamline isolation, has been experienced during other reactor trips (e.g., Unit 2 LER 311/90-029-00). The failure of 21TB10 exacerbated the cooldown. Engineering has investigated T sub avg reduction (during trips) and design modifications are being assessed.

CORRECTIVE ACTION:

Operations procedure IOP-3, "Hot Standby to Minimum Load" has been revised to not allow reactor power to exceed 8% if S/G level control is maintained solely on the operation of the BF40 valve. This was verified via a Salem Simulator test run which showed that the BF40 will not maintain mass inventory above 8% reactor power.

The circumstances surrounding this event will be reviewed with applicable Operations Department personnel.

The circumstances surrounding this event will be reviewed by the Nuclear Training Center for incorporation into applicable training programs.

PSE&G will review its past response to INPO Significant Operating Experience Report No. 84-4, "Reactor Trips Caused By Main Feedwater Control Problems". This review will include review of this event as well as past reactor trip events, which occurred on transients involving S/G level, steam flow or feed flow.

The BF19 and BF40 valves position indication will be reviewed and appropriate actions taken.

Repairs were made to the No. 23 S/G feedwater control system.

The 21TB10 valve positioner linkage arm was repaired.

The CBD development process will be assessed to determine if revised

CBD information is appropriately disseminated and used.

General Manager -
Salem Operations

MJP:pc
SORC Mtg. 92-066

ATTACHMENT 1 TO 9206230187 PAGE 1 OF 1

PSE&G

Public Service Electric and Gas Company
P.O. Box 236 Hancocks Bridge, New Jersey 08038

Salem Generating Station

June 12, 1992

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Dear Sir:

SALEM GENERATING STATION
LICENSE NO. DPR-75
DOCKET NO. 50-311
UNIT NO. 2

LICENSEE EVENT REPORT 92-009-00

This Licensee Event Report is being submitted pursuant to the requirements of the Code of Federal Regulations 10CFR 50.73 (a)(2)(iv). This report is required to be issued within thirty (30) days of event discovery.

Sincerely yours,

C. A. Vondra
General Manager -
Salem Operations

MJP:pc

Distribution

The Energy People

*** END OF DOCUMENT ***